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DESCRIPTION OF NEUTRON ATTENUATION BY BIOLOGICAL SHIELD WITH THE AID OF MULTIGROUP MODEL TO DIFFUSION APPROXIMATION

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INTRODUCTION

Recently, in some countries a large number of works is carried out in which nuclear reactors are used as power source in both great stationary power plants and transport units. Reactors of such plants are powerful radiation sources because of high thermal stress in their cores, To attenuate the radiation to the level which is safe for the human health, the thickness of a biological shield may be of several meters and its weight may reach some hundreds of tons. The main demand of the reactor radiation shield, for example, of a transport plant, is threferore that for its minimum weight and size, and the design of some plants with atomic engines depends completely on the solution of that problem. The experience in designing of great stationary atomic power plants has revealed the importance of shielding since costs of a plant depend essentially on the shield weight. Designing of a nuclear reactor radiation shield of moderate weights and sizes is extremely difficult, and it may be successfully fulfilled only when a designer has sufficiently reliable and simple calculation methods for reactor-radiation attenuation.

In the present work, the author attempts to extend multigroup methods of the diffusion theory, used for calculation of physical effects in the reactor core, to calculation of neutron-flux attenuation by a multilayer biological shield and to prove their applicability to metal-water shields. The applicability of the present method to calculation of thermal and biological shield is shown by way of the analysis of predicted and experimental data on the neutron-flux distribution in biological shields of operating reactors of the research type-IRT (U.S.S.R.) and the power type -APPR (USA).

METHOD

Main assumptions and equations. The simpliest form of neutron-flux equations may be found by using the diffusion approximation of the kinetic equation which is widely adopted in nuclear-plant designing. Several workers have applied the diffusion approximation to the development of calculation method of neutron flux also [1], but the analysis involving assumptions of the diffusion theory was, as a rule, restricted to slowing-down and thermal neutrons solely. Moreover, fast neutron attenuation was prescribed by a certain semi-theoretical method.

In the present work the possibility is shown to describe the behaviour of both thermal and slowing-down as well as fast neutron in the framework of the diffusion theory. The solution is based on a number of peculiarities of the neutron-flux attenuation mechanism in a water-metal shield. These are:

(a) Fast neutron are essentially attenuated by inelastic collision with heavy nuclei (Fe, Pb). Inelastic scattering may be considered isotropic to the first approximation.

(b) Fast neutrons of high energy of the order of 4-6 Mev are very probable to excite the first levels of the medium nuclei (Fe, Ni, Cr etc.) with a relatively small energy loss. This mechanism of neutron interaction is similar to elastic scattering.

(c) Though extremely anisotropic on all nuclei at high neutron energies, elastic scattering is inessential for the attenuation mechanism and reactions of fast neutron absorbtion seem hardly possible.

(d) In the range of energies below inelastic scattering threshold for the nuclei of Fe, Pb, elastic scattering, isotropic or slightly anisotropic on all nuclei (except hydrogen) is the main attenuation mechanism.

The above factors provide promises for application of the spherical harmonic method to low approximations to calculation of attenuation of fast (and especially slowing-down) neutrons when angular neutron distribution is characterized by a small number of series terms with respect to spherical functions, in particular in the diffusion approximation, by two terms. In addition because of rough representation of angular neutron distribution passing long distances inside hydrogen-containing shield, it should be expected that this method will cause worsening of the accuracy

with the distance from the source.

A biological shield may be considered to a first approximation to consist of a number of successive uniform media. Let us consider such a medium, assuming that:

1. The medium absorbs and scatters neutrons.

2. Inelestic scattering is isotropic.

3. According to Weiskopf's statistical theory, neutron spectrum inelestically scattered on heavy nuclei (Pb, U-235, U-238) has the form of Maxwellian distribution. The inelastic scattering law for medium nuclei (Al, Fe, Ni, Cr) is obtained by excitation functions of nuclear levels predicted by means of assumptions of the optical nuclear model [2].

4. Elastic scattering is non-isotropic in the coordinate system where the centre of the neutron mass and scattering nucleus is fixed. Scattering anisotropy is estimated by linear and angular scattering law.

Then, in accordance with the method, the scalar neutron flux should satisfy (as the function of space and energy), the following equation for the steady-state case:

$$-D(E)\Delta\phi(\vec{r},E)+\Sigma(E)\phi(\vec{r},E)=\int_E^{\infty}\phi(\vec{r},E')\Sigma_s(E')W_{s0}(E',E)dE'+\int_E^{\infty}\phi(\vec{r},E')\Sigma_{in}(E')W_{in_0}(E',E)dE' \quad (1)$$

with boundary condition corresponding to the real geometry. In equation (1) W_{s0}, W_{in0} are coefficients of indicatrices expansion of elastic W_s and inelastic W_{in} scattering which are equal to

$$W_{s0}(E',E)=\frac{(A+1)^2}{4AE'}\left\{1+\frac{3}{2}\frac{A^2}{A^2-0{,}6}\left(\bar{\mu}_L-\frac{2}{3A}\right)\left[(A+1)^2\frac{E}{E'}-(A^2+1)\right]\right\}$$

$$W_{in_0}(E',E)=4\pi\cdot W_{in}=E\cdot\exp\left[-\frac{E}{T(E')}\right]\left\{T^2(E')\left[1-\left(1+\frac{E}{T(E')}\right)\exp\left(-\frac{E}{T(E')}\right)\right]\right\}^{-1}$$

and the indicatrix of elastic scattering

$$W_s(\mu_0,E',E)=\frac{(A+1)^2}{16\pi AE'}\left\{1+\frac{3}{2}\frac{A^2}{A^2-0{,}6}\left(\bar{\mu}_L-\frac{2}{3A}\right)\left[(A+1)^2\frac{E}{E'}-(A^2+1)\right]\right\}\delta\left[\mu_0-\left(\frac{A+1}{2}\sqrt{\frac{E}{E'}}-\frac{A-1}{2}\sqrt{\frac{E'}{E}}\right)\right]$$

for $\alpha E'<E<E'$, μ_0 is the cosine of the angle between neutron directions before and after scattering, $\bar{\mu}_L$ is the mean cosine of the scattering angle in the laboratory coordinate system. The rest designations are usual.

The solution of equation (1) for the real shield geometry involves great difficulties because of complex dependances of neutron cross-sections and indicatrices of elastic and inelastic scattering on energy. The integro-differential equation which contains variables - space coordinates and energy - is therefore reduced to the system of connected differential multigroup equat-

ions by means of integration with respect to chosen energy intervals. Each equation of the system describes the behaviour of the neutron flux (integral) ϕ_j of the infinite energy interval ΔE_j in a layer of the shield. It is written as follows:

$$\beta_j \frac{d\phi_j(z)}{dz} - \Sigma_j \phi_j(z) + \sum_{\kappa=1}^{j-1} \phi_\kappa(z) \Sigma_{\kappa j}^{in} + \sum_{\kappa=1}^{j-1} \phi_\kappa(z) \Sigma_{\kappa j}^{s} = 0$$

$$\Sigma_j = \Sigma_{aj} + \sum_{\kappa=j+1}^{N} (\Sigma_{j\kappa}^{in} + \Sigma_{j\kappa}^{s}); \quad \Sigma_{j\kappa}^{in,s} = \int_{\Delta E_j} dE \int_{\Delta E_\kappa} \phi(z,E') \Sigma_{in,s}(E') W_{in,s}(E', E) dE' / \phi_j$$

$$\Sigma_{aj} = \int_{\Delta E_j} \Sigma_a(E) \phi(z,E) dE / \phi_j; \quad \beta_j = \int_{\Delta E_j} \beta(E) \nabla\phi(z,E) dE / \int_{\Delta E_j} \nabla\phi(z,E) dE \qquad (2)$$

As we can see from the above formula, the group constants are not constant, but they depend on the unknown space-energy neutron-flux distribution and cannot be estimated if the wanted solution is unknown. To overcome this difficulty, for estimation of the constants, spectra were used which are formed in infinite media with uniformly distributed fission neutron sources.

Choise of groups. The main aim of the division of the energy interval was the description in every detail main processes effecting penetration of neutrons through the layer of metal-water biological shield. The following factors were accounted for: the dependence of neutron cross-sections on energy for main shielding materials (water, iron, lead), the functions for conversion of the neutron flux into the biological tissue dose, as well as neutron spectrum shape in uniform infinite media of water, iron, lead and graphite. In accordance with the dependence of the above quantities on the neutron energy, we may divide the whole range of energy into several great regions.

1. From 1.5 to 15 Mev where the flux shape follows the fission spectrum in the majority of the considered media. This region is characterized by great anisotropy of elastic scattering of neutrons and attenuation mainly due to inelastic scattering in the media containing heavy nuclei. Here three groups are formed: j=1 4 Mev < E < 15 Mev, j=2 2.5 Mev < E < 4.0 Mev, j=3 1.5 Mev < E < 2.5 Mev. In all these groups total cross-section of elestic scattering materials-oxygen and carbon, cross-sections of aluminium, iron, nickle, lead and uranium as well as the conversion factor for the neutron flux into the tissue dose depend slightly on energy. Hydrogen is the only exception of which cross-section changes in the first group by factor 2 and in second group within 40 per cent.

2. In the region of transition from the fission spectrum to the Fermi spectrum (above 0.3 Mev) two groups are distinguished:

j=4 0.7 Mev < E < 1.5 Mev, j=5 0.3 Mev < E < 0.7 Mev. In this group inelastic scattering is of some importance for neutron flux attenuation in iron, lead and uranium, but in lighter media elastic scattring (which is non-isotropic in the centre of mass) is the dominant process.

3. The region of the Fermi spectrum from 1 to 300 kev, in which characteristic features of neutron interaction are isotropic elastic scattering, interference between potential and resonant scattering and slight neutron absorbtion. The resonant structure of the total cross-section due to interference should be carefully accounted for in averaging cross-section. Presence of dips in the total cross-section leads to the fact that neutron after attenuation may attain the energy close to the dip energy and its path in the shield may be long. To account for this effect two groups are distinguished: j=6 40 kev < E < 300 kev, j=7 1 kev < E < 40 kev.

4. The region from 0 to 1 kev where neutron absorbtion is essential for a lrage number of shielding materials, three groups are chosen: j=8 6.7 ev < E < 1000 ev, j=9 $E_{rp.}$ < E < 6.7 ev, j=10, 0 < E < $E_{rp.}$

Thus, 10 energy groups are distinguished. The last group includes thermal neutrons. Attenuation of thermal neutron flux is characterized by effective cross-sections which are averaged following the Maxwellian spectrum [3]. Conventional division of thermal and slowing-down neutrons by the value $E_{rp.}$ is taken from the method applied to calculation of reactor critical masses [3].

THEORETICAL AND EXPERIMENTAL DATA ON NEUTRON FLUX ATTENUATION IN UNIFORM MEDIA. CRITERION OF METHOD

A great number of assumptions used for derivation multigroup diffusion equations restricts the validity of the method and demands theoretical and, especially, experimental verification of the method. Theoretical verification of its applicability to calculations of neutron-flux attenuation in a biological shield seems practically impossible, since it involves determination of spatial and angular and energy distribution of neutrons in a multilayer medium in the region of distributed fission sources. The analysis of neutron flux energy spectra formed in

infinite uniform media by a lumped source is easier.In addition the analysis of neutron attenuation by various uniform media allows distinguishing the errfors due to assumptions of the diffusion theory from the errors caused by inaccurate description of interaction between the media in formation of neutron flux taking place in a uniform shield.Accurate energy spectra of the flux in homogeneous media calculated by the momentum method [4] were therfore used as the criterion of the correct results obtained by the diffusion ten-group method.The data on neutron attenuation by various indicators (copper-63,indium-115,sulphur-32,thorium-232) and some others may be also used as the criterion.Materials widely used in designing of metal-water biological shieldings may be conventionally divided into three classes according to the type of interaction between fission neutrons with atom nuclei:

1.Media attenuating neutrons by elastic scattering with a rather small contribution of inelastic scattering.

2.Media where fast neutrons are attenuated mainly by inelastic scattering.

3.Media containing hydrogen.

The most typical shielding materials (graphite,lead,iron and water) were considered in accordance with this division.

Graphite. Neutrons of the fission spectrum interact with graphite through elastic scattering since the first level of the carbon nucleus has the energy of about 4.4 Mev.Graphite is the medium which slightly attenuates neutrons,and the errors in the flux should be threfore caused by inaccurate representation of the angular neutron flux distribution and elastic scattering indicatrix.The latter inaccuracy arising in the case of anisotropic scattering in the system of the centre-of-masses leads to underestimation of the contribution to the neutron flux of neutrons scattered with small angles.The experience has shown [3] that the assumption on a linear scattering indicatrix is rough for neutrons of the fission spectrum of which energies exceed 2.5 Mev,especially,in the energy range above the firs level of a carbon nucleus.Thus,essential errors may be expected both in the neutron-flux spectrum for $E > 4.0$ Mev and in the low-energy region of the spectrum at the distances from the source where the flux is determined completely by neutrons attenuating

from region E 4.0 Mev.The order of the value of the distance where the first flux errors should be expected,may be estimated for the case of a plane isotropic fission source for which group fluxes (solutions of equation (2)) are determined by the following recurrent relations:

$$\Phi_j = \sum_{k=j}^{j} A_k \alpha_{kj} \exp(-\varkappa_k x) \; ; \quad \alpha_{kj} = \sum_{m=k}^{j-1} \varkappa_{mj}^2 \alpha_{km} / (\varkappa_j^2 - \varkappa_k^2) \; ; \quad \alpha_{jj} = 1 \; ;$$
$$A_k = \left[\frac{\chi_k}{2D_k} - \sum_{m=1}^{k-1} \varkappa_m \alpha_{mk} A_m \right] / \varkappa_k \; . \qquad (3)$$

where x is the distance from the source, $\varkappa_j^2 = \Sigma_j / D_j$;and $\varkappa_{kj}^2 = \Sigma_{kj} / D_j$.

For graphite of the density 1670 kg/m^3 $\varkappa_1$=8.00; $\varkappa_2$=12.2; $\varkappa_3$=12.6 m^{-1},D_1=0.0543,D_2=0.0306,D_3=0.0265 m. χ is the fission spectrum.

It is easy to see that the neutron flux of the first group exceeds essentially (say,by one order) the neutron flux from the source in the range 1.5 Mev<E<4 Mev beginning from the distance

$$x = \left\{ \ln \left[10 \frac{\varkappa_1}{\varkappa_2} \frac{D_1}{\chi_1} \left(\frac{\chi_2}{D_2} + \frac{\chi_3}{D_3} \right) \right] \right\} / (\varkappa_2 - \varkappa_1) \; , \quad x \approx 0{,}85 \text{ m} .$$

Enegry distribution of the neutron flux predicted by equation (3) for the distances γx =10,60,90,120 g/cm^2 was compared with the appopriate values calculated by the momentum method [4], see Fig.1.The agreement of absolute energy spectra at the distances up to 120 g/cm^2 was fairly well.The maximum error of about 40 per cent appears at the distance x=0.72 m from the source in a narrow specrum range 0.7 Mev<E<2.5 Mev.These data confirm the above assumptions and show the limits of the validity of the ten-group method for calculation of neutron flux attenuation in graphite with the allowable accuracy with the aim of shielding.

Iron,lead. In iron and lead attenuation of fast neutrons is mainly defined by inelastic scattering.It is more important to properly determine attenuation of fast neutrons by iron and lead with energies above 4 Mev and resonance neutrons with energies below a threshold of inelastic sxattering E<0.85 Mev. If fast neutrons with E>4 Mev are relatively slightly moderated with water,and therefore, make a considerable contribution to the dose rate in metal-water shielding,then the resonance neutrons are more penetrating in lead and iron;they cause the flux of thermal neutrons in water layers following lead and iron in the lead-water or iron-water shieldings.

A space-energy distribution of the neutron flux in an infinite massive Fe with a density of 7.85 g/cm^3 for the case of a point isotropic fission source of the power 1 neutron/sec is

calculated to verify group iron constants by the ten-group method.The measurement results on attenuation of neutrons in iron which are emitted by a monodirected beam from the reactor BR-5 [5]are most similar to be compared with the predicted values.The distribution of fast neutrons from the monodirected beam from the reactor on the fast neutrons BR-5 is measured by threshold indicators $Al^{27}(n,\alpha)Na^{24}$, $S^{32}(n,p)P^{32}$ and the thorium fission chamber.The results of experiments with monodirected source of reactor neutrons can be compared with the predicted values of the present work (See Fig.2a),since the attenuation function of an infinite flat monodirected source and that of a point isotropic source multiplied by $4\pi r^2$ do not practically differ [7] and the spectrum of the fast neutron reactor in the region above the lowest threshold of reactions occuring in the used detectors (reactions of Th (n,f)) is close to the fission spectrum. The calculated activities of the sulphuric indicators,the thorium chamber fissions and the corresponding measured values agree satisfactorily at distances up to 0.6 m from the source. The 30 per cent maximum difference is observed in the thorium chamber fissions at r=0,6 m.The similar comparison made for counting the thorium chamber from the point isotropic source 4 Mev of neutrons in lead at distances up to 0.52 m from the source (See Fig.2b) has shown that the maximum difference between the predicted and measured data[6] is 20 per cent.The comparisons carried out show that the 10-group diffusion method may be used for calculation of both fast and resonance neutrons in iron and lead at distances up to 0.6 m from the source.

Hydrogen-containing media.Due to the fact that hydrogen scattering on nuclei is spherically symmetric in the centre-of-mass system,moderation of neutrons by hydrogen in diffusion balance equation (1) is defined presisely,and the diffusion theory must be due to a linear representation of the angular dependence of the neutron density,from one side,and the use of the diffusion law,from the other side.The latter assumption means that a neutron scattered by hydrogen nucleus changes the direction of its motion by the constant angle arc cos 2/3.But in reality the neutron changes its direction depending on the fact to which energy group it is moderated and for the fission

neutrons the averaged within the groups coefficient $W_{s_1}(E,E') = \frac{1}{E'}\sqrt{\frac{E}{E'}}$ is less than 2/3. It may therefore be assumed that in the case of the diffusion description, the neutron pass longer paths in a medium containing hydrogen than in reality. The effect of scattering on hydrogen should lead to the overestimation of the fast neutron flux at distances from the source, where the anisotropic angular distribution is not formed. If it is so, then the increase in the approximation order, as before without exact account for correlation between changes in angle and energy with scattering should to worse results than in the diffusion approximation. For example, while describing the neutron attenuation in the P_2 approximation, in this case the balance equation has the form which coincides with that of the diffusion equation and differs only by constants [8]. In the case of pure hydrogen, the diffusion length in the P_2 approximation for the first group of the most penetrating neutrons is expressed through the constants in the diffusion approximation and through the mean square of the path with respect to scattering as follows $L^2_{P_2} = \frac{D}{\Sigma_1} + \frac{16}{15}\left(\frac{1}{\Sigma_{s_1}}\right)^2$.

The predicted diffusion length in the various approximations are depicted in Table I. From the Table it is seen that the use of the subsequent P_1 and P_2 approximations in the discussed form will lead to an unjustified strong attenuation in the P_1 approximation and to more weak in the P_2 approximation than it occurs in reality.

The presence of heavy elastic scatterer in a hydrogen-containing medium (for instance, oxygen in water) which do not moderate the neutrons as compared to hydrogen but compete with it during the scattering process, should improve the results of the diffusion methods due to underestimation of scattering by small angles, and consequently, due to underestimation of the transport paths of neutrons. The overestimation and underestimation of the transport path while describing the moderation on hydrogen and oxygen lead to compensation of diffusion-approximation errors. This considerably explains a good agreement of absolute energy spectra in water at a distance up to 0.9 m from a point fission source which are calculated by the ten group and momentum methods (see Fig.3a). In addition, at a distance of 0.9m from the source, the differences in fluxes predicted according to these two methods in the range 0.3 Mev<E<8 Mev does not exceed 30 per cent. The predicted values for the indium activa-

tion distribution and the similar experimental data for water [9], [11] (Fig.3b) are in the same quantitative ratio.

APPLICATION OF METHOD FOR CALCULATING THERMAL REACTOR VESSEL SHIELDING AND BIOLOGICAL SHIELDING

The calculation precision in a real heterogeneous shielding of such quantities as the flux of thermal and intermediate neutrons, dose rate by neutrons and capture γ-radiation sources depends on the correct description of fast neutron attenuation in the majority of cases which are important in practice. Exceptions are thick lead or steel slabs which are sometimes used as protective shields or elements of the reactor construction. In such media the total neutron flux, the dose rate by neutrons and capture γ-radiation sources should be defined by resonance neutrons. It is very important to take into account the accumulation of resonance neutrons in metals while calculating the capture γ-radiation sources in a metal-water shielding. To obtain the correct information on attenuation of resonance neutrons by metal protective shield, it is necessary to correspondingly choose a way for averaging group constants.

While the resonance neutron flux in thick shield made of nickle, iron and lead with sizes of some paths is calculated, the interaction cross-sections should be averaged by the Fermi spectrum, in thin shields of the metal-water shielding, by the spectrum $\phi(u)=\text{const}$, as in the latter case the spectrum is formed by hydrogen where in the resonance region E=const. In this respect iron is more typical as one of its isotops has a deep dip at 24 kev in the cross-section. The diffusion length for neutrons of the seventh group containing a dip in the scattering section which are calculated by group constants acording to the ways of averaging prove to be equal respectively for the spectrum $\phi(u) \sim 1/\Sigma(u)$ $L=\sqrt{D/\Sigma}=0.361_m$ and for $\phi(u)=\text{const}$ L=0.16 m. The first quantity agrees well with the relaxation lenth of intemediate neutrons measured by Broder [6] in iron, and equal to 0.36 m, the second is close to the moderation length of intermediate neutrons in iron of the iron-water shield L=0.177 m obtained by Cooper et al, by the matching of perdicted data for flux to the experimental ones [1]. The group sections for metals averaged by two types of a spectrum are given in Table II.

The satisfactory data obtained by calculation of fast neut-

ron attenuation in uniform media provide promisis for calculating the space-energetic distribution of the neutron flux in a multilayer biological shielding at some distance from the core by means of the 10-group method.

The possibility for using the 10-group method to calculate the flux of thermal and slowing-dow neutron in thin water interlayers in strongly absorbing media and near such media is referred to problems on the applicability of the method for perdicting attenuation by the heterogeneous shielding.

The work on the analysis of calculation and measurement results of the neutron flux in shield assemblies initiating thermal vessel shielding, vessel and a part of the biological shielding of a nuclear reactor [10] is devoted to verify these effects. The comparison of calculations of the neutron flux by copper indicator activity with the experimental results allows the following conclusions: 1. the ten-group system of constants satisfactorily describes the behaviour of a neutron flux in steel, steel borated and lead shield of the thermal vessel shielding. The coincidence of the predicted and measured activities within 20-30 per cent may serve as a basis to judge about the applicability of the method to calculating the capture gamma-radiation sources in iron-water and lead-water shields up to 0.4 m in thickness; 2. the maximum difference between the predicted and measured quantities is observed in media containing hydrogen between borated shield-strong absorbers- and reaches 30 - 40%. These deviations may be due the asymmetry of the angular neutron distributions in layers which absorb neutrons strongly as well as when a real spectrum of a neutron flux in these layers differs from the assumed one while averaging the group constants.

Great possibilities of the 10-group method are illustrated by comparison of the experimental data on measurement of neutron flux distribution by golden indicators and the dose rate by γ-rays in the iron-water biological shield of the American reactor APPR [12] and the thermal neutron flux in a thermal graphite column of the research pool reactor IRT-2000 [13] with the predicted values. The comparison of the measured and predicted activities of the golden indicators in the biological shielding of APPR has shown that the 10-group method satisfactorily describes the neutron flux attenuation in the biological shielding up to 1 m, i.e. in the region which causes the dose rate outside

the biological shielding.The distribution of the capture γ -radiation sources in the biological shielding of APPR was calculated on the basis of the space energy neutron-flux distribution by the 10-group method,and then the distribution of the dose rate (See Fig.4),by the constructed functions of the dose.The predicted absolute total dose rate coincides with the measured one at distances up to 0.4 m from the centre of the core within 30 per cent (only at one point of the shielding the predicted dose rate differs from the measured one by factror 1.5).

Neutron flux attenuations in the thermal column of the rector IRT-2000 werecomputed on electronic computer Ural-1, at the Institute of Mathematics and Computer Technique of the BSSR Academy of Soiences.The comparison of the results obtained with the experimental data (See Fig.1) shows a satisfactory precision of calculation of the flux of thermal and slowing-down neutron in graphite.This work illustrates one of the main advantages of the 10-group method,i.e. the simplicity of programming of the algorithm method even on such low-class computers (by "memory") as Ural-1.

In conclusion it should be noted that despite the satisfactory agreement of predicted and experimental data for the sake of shielding the certain caution is required for the application of the diffusion multigroup methods (especially in cases which are not verified by experiments) due to a rough representation as well as to the scattering indicatrix.

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721

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721

Table I

Predicted Diffusion(Relaxation) Lengths, mm for Hydrogen of Density 111 kg/m^3

approximation / interval	Diffusion	P_1	P_2	Momentum method [4]
$8<E<\infty$	0.157		0.22	
$4<E<\infty$	0.1105	0.0628	0.142	
$7.9<E<8.1$	0.130	0.0772	0.190	0.132

Table II

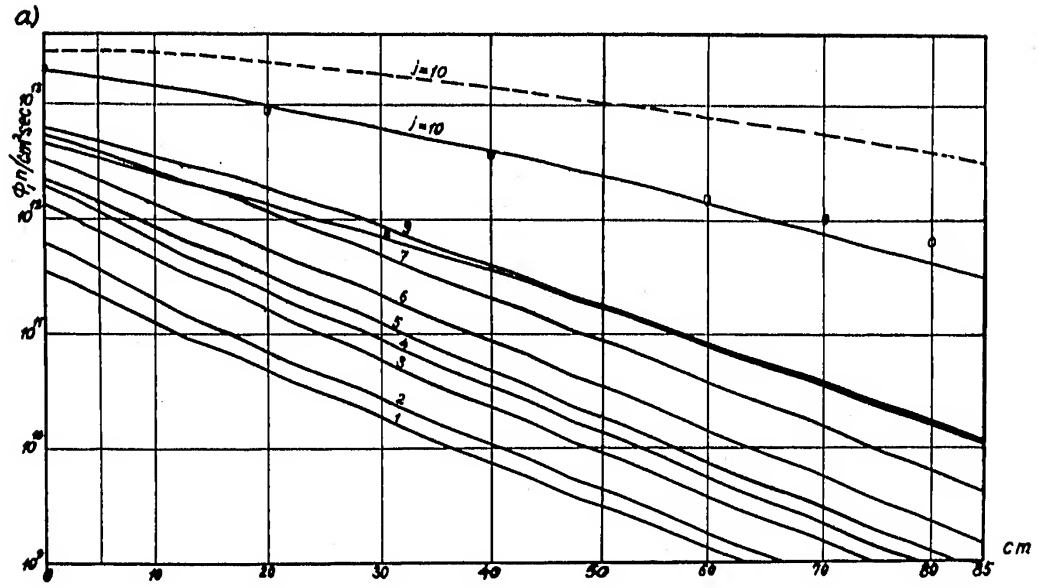
Group Diffusion Lengths for Some Materials

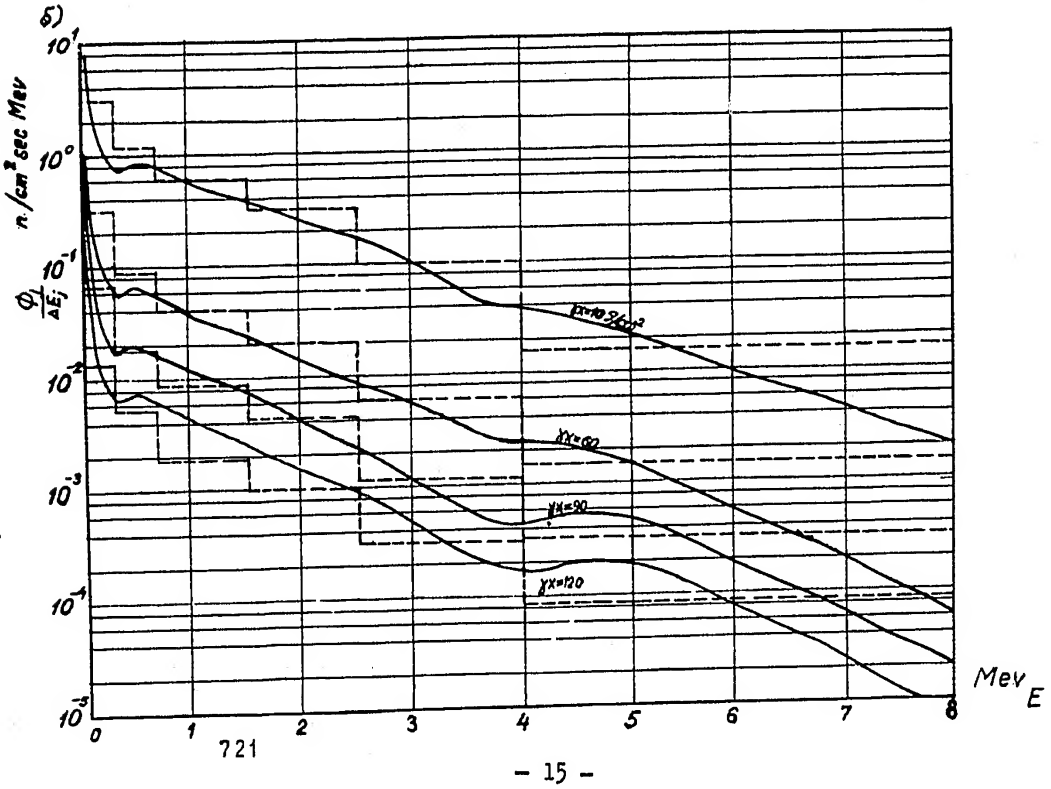
L_a-constants in group 5-9 are averaged by the spectrum $\phi(u)=1/\Sigma(u)$, L_b-by the spectrum $\phi(u)=\text{const}$.

j	Material density	Iron, 7850		Lead 1130		Water 1000	Graphite 1670
	E_{min_j} ev	L_a	L_b	L_a	L_b	L_b	L_b
1	$4\cdot10^6$	0.0521	0.0521	0.0633	0.0633	0.0752	0.125
2	$2.5\cdot10^6$	0.0461	0.0461	0.0789	0.0789	0.0545	0.082
3	$1.5\cdot10^6$	0.0532	0.0532	0.108	0.108	0.0447	0.0793
4	$0.7\cdot10^6$	0.0803	0.0803	0.172	0.172	0.0282	0.0653
5	$0.3\cdot10^6$	0.129	0.121	0.255	0.238	0.0210	0.0506
6	$40\cdot10^3$	0.209	0.179	0.256	0.246	0.0164	0.0606
7	$1\cdot10^3$	0.361	0.164	0.249	0.292	0.0134	0.0729
8	6.7	0.0662	0.0576	0.33	0.320	0.0145	0.0838
9	0.189	0.0372	0.0314	0.276	0.218	0.0114	0.0717

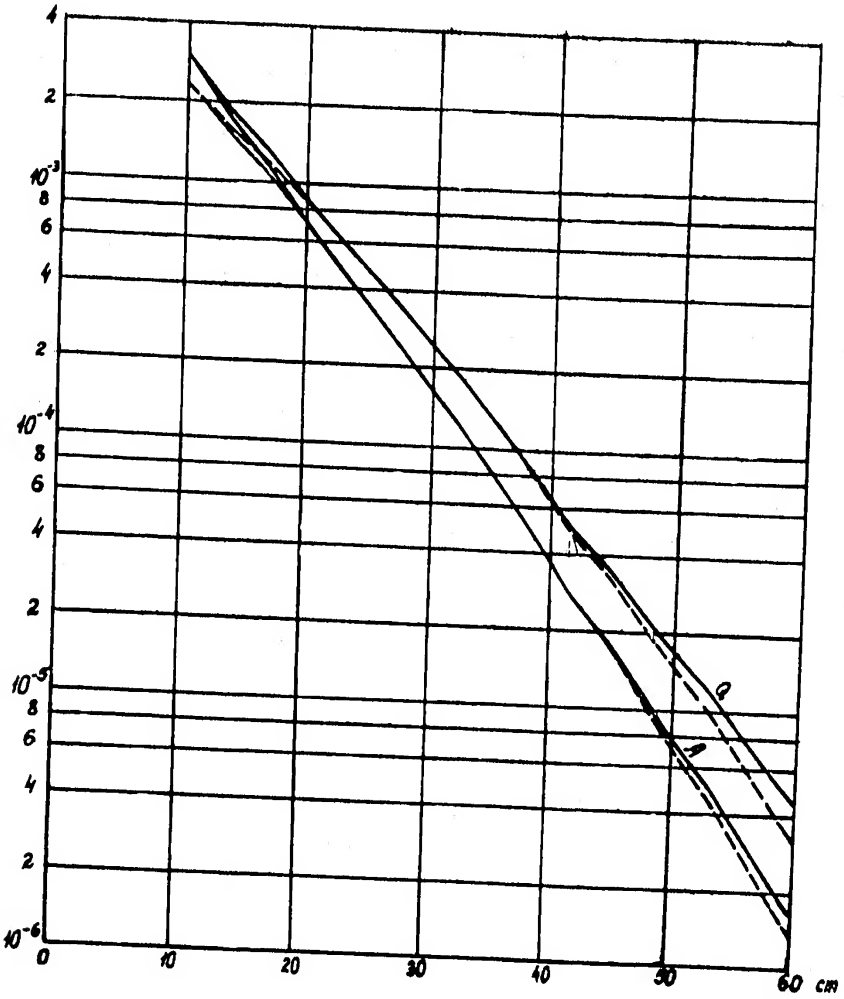
721

Fig.1.





а)



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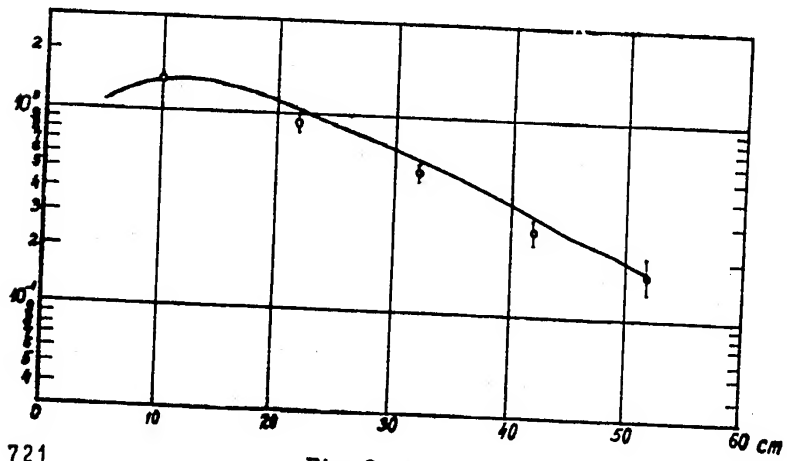

721

Fig.2.

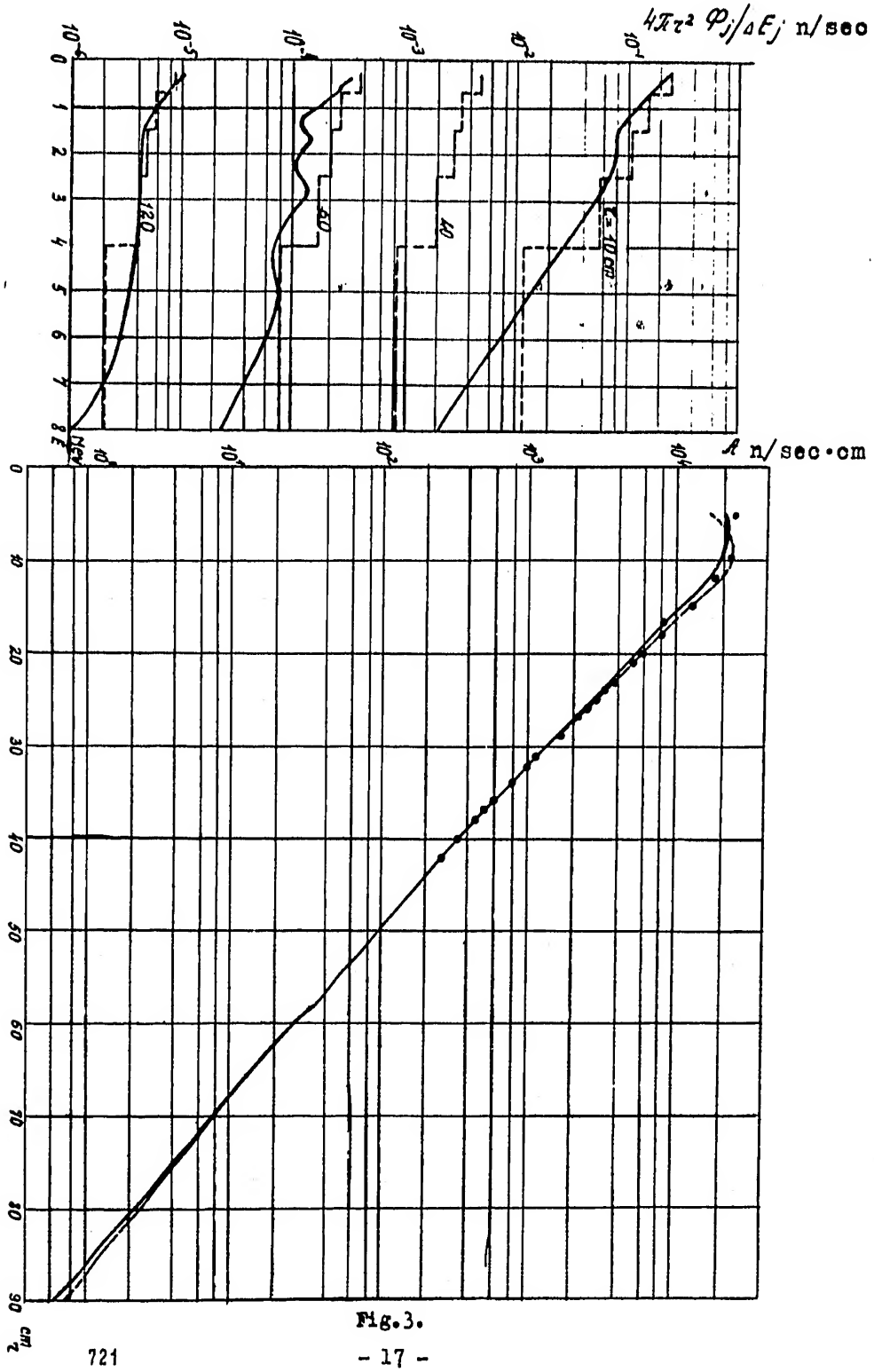

Fig.3.

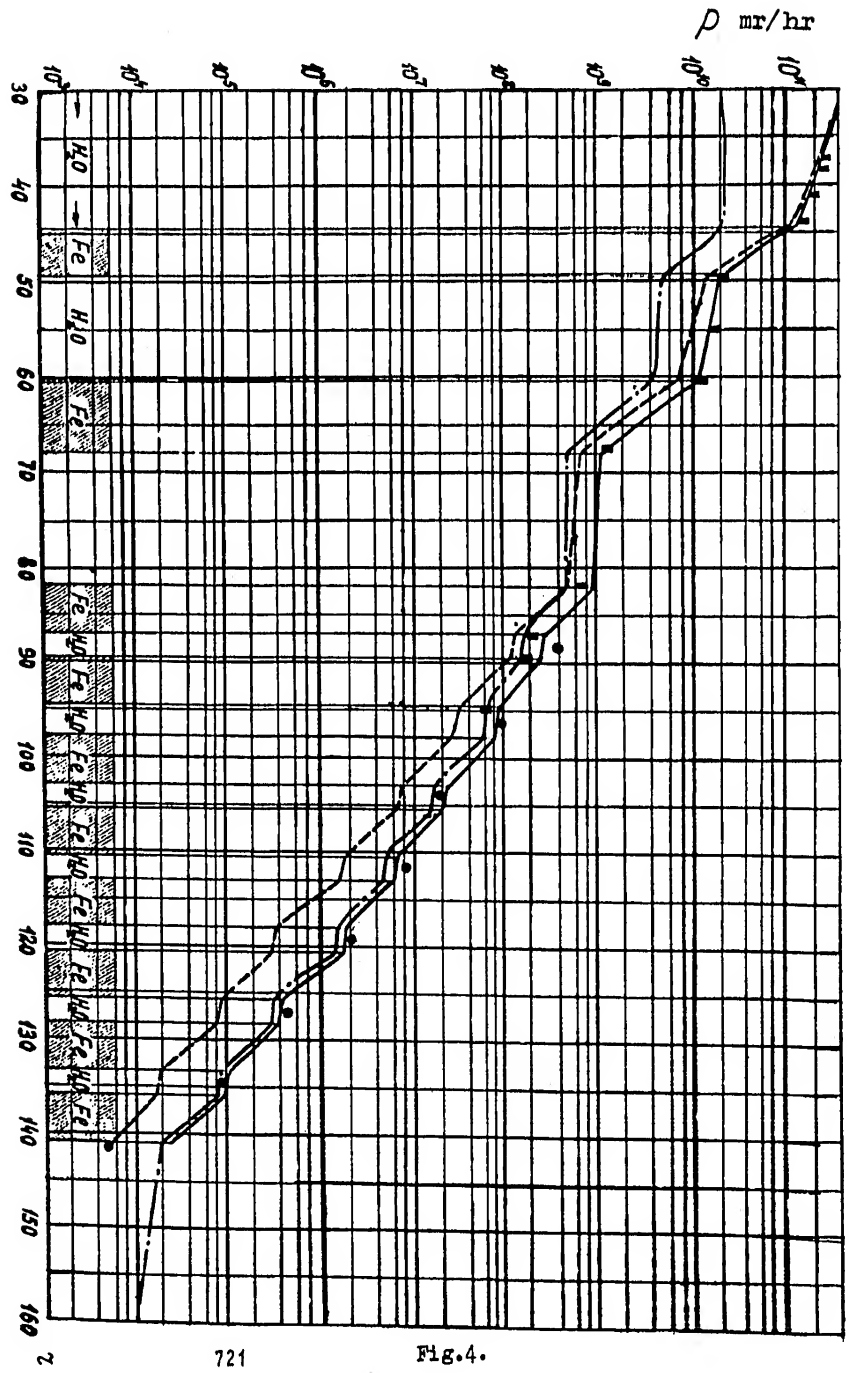

Fig.4.

FIGURES

Fig.1. Neutron attenuation in graphite.

Fig.2.Comparison of predicted (point isotropic source) and experimental data on neutron attenuation in iron and lead.

Fig.3. Water attenuation of neutron flux from point isotropic source (of unit power) of fission spectrum.

Fig.5 Dose rate in thermal shield of vessel and biological shield of the APPR reactor P mr/hr.

721